

NON-PUBLIC?: N
ACCESSION #: 9406020024
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Washington Nuclear Power Plant - Unit 2 PAGE: 1 OF 6

DOCKET NUMBER: 05000397

TITLE: MANUAL SCRAM DUE TO OBSERVED REACTOR CORE POWER
FLUCTUATIONS

EVENT DATE: 4/26/94 LER #: 94-008-00 REPORT DATE: 05/26/94

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 50

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: C.D. Mackaman, Licensing Engineer TELEPHONE: (509) 377-4451

COMPONENT FAILURE DESCRIPTION:

CAUSE: BD SYSTEM:

AD COMPONENT: ZT MANUFACTURER: S052

REPORTABLE NPRDS: NO

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 1010 hours on April 26, 1994, with the plant at 50% reactor power and 55% core flow, plant Control Room Operators (CROs) observed slight reactor power fluctuations on the six average power range monitors (APRMs). Based on an indication of potential core instabilities, Control Room personnel manually scrammed the reactor in accordance with Plant Abnormal Operating Procedure PPM 4.12.4.7, "Unintentional Entry Into Region of Potential Core Power Instabilities."

Immediate corrective actions were taken by the Control Room staff to bring the plant to a safe shutdown condition in accordance with Emergency Operating Procedure (EOP) 5.1.1, "RPV Control," and Recovery Procedure PPM 3.3.1, "Reactor Scram."

The root causes for this event were the degraded condition of a Reactor

Recirculation (RRC) flow control valve (FCV) position transmitter and the lack of procedural criteria to determine when to "lock up" the degraded FCV.

Further corrective actions include: (1) replacement of the RRC FCV position transmitters, (2) calibration and testing of the replacement FCV position transmitters, and (3) required reading of this Licensee Event Report (LER) by WNP-2 licensed operators.

This event posed no threat to the health and safety of either the public or plant personnel.

END OF ABSTRACT

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Plant Conditions

Power Level - 50%

Plant Mode - 1 (Power Operation)

Event Description

At 0831 hours on April 26, 1994, WNP-2 experienced an overload trip of a 480 VAC load center breaker (E-CB-31/3) and loss of a nonsafety-related 480 VAC motor control center (E-MC-3A). The loss of the motor control center (MCC) caused a loss of power to the bleed steam (BS) dump and turbine non-return (backflow preventer) valve solenoid pilot valves for Low Pressure (LP) Feedwater Heaters 1A, 2A, 3A, and 4A. The loss of power to the solenoid pilot valves resulted in isolation of extraction steam to the four LP heaters and opening of the dump valves to the main condenser. By 0841 hours, reactor feedwater (RFW) inlet temperature had decreased approximately 6.5 degrees Fahrenheit. At 0850 hours, in accordance with Plant Abnormal Operating Procedure PPM 4.2.7.2, "Loss Of Feedwater Heating," plant Control Room Operators (CROs) reduced reactor power from 70% to 50% and core flow from 100% to 55%.

At 1010 hours, plant CROs observed slight reactor power fluctuations on the six average power range monitors (APRMs) during an approximate four minute time period. The maximum amplitude of the fluctuations was approximately 8% peak-to-peak and the interval between fluctuations was approximately 20 seconds. Based on the increasing amplitude of the fluctuations, Control Room personnel manually scrambled the reactor less than one minute after the maximum amplitude was experienced using guidance outlined in Plant Abnormal Operating Procedure PPM 4.12.4.7, "Unintentional Entry Into Region of Potential Core Power Instabilities."

All control rods fully inserted and no safety relief valves actuated. Plant response to the scram was as expected. The reactor pressure vessel (RPV) level reached a minimum of -15.4 inches and a maximum of +60.1 inches.

Immediate Corrective Actions

Following the reactor scram, the Control Room staff promptly entered Emergency Operating Procedure RPV level was (EOP) 5.1.1, "RPV Control, as required when the RPV level decreased to +13 inches recovered using the RFW pumps and the plant was stabilized in accordance with Recovery Procedure PPM 3.3.1, "Reactor Scram." EOP 5.1.1 was excited at 1043 hours.

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Further Evaluation and Corrective Action

Further valuation

1. Pursuant to 10CFR50.72(b)(2)(ii), this event was reported to the NRC Operations Center via the Emergency Notification System (ENS) at 1111 hours as an unplanned manual actuation of the Reactor Protection System (RPS). This event is also being reported in accordance with 10CFR50.73(a)(2)(iv) as an unplanned manual actuation of the RPS.
2. The overload trip of Load Center Breaker E-CB-31/3 and loss of Motor Control Center E-MC-3A occurred while performing a test run of a new turbine building exhaust fan (TEA-FN-1C) motor with it coupled to the fan. The motor had been upgraded from 100 HP to 200 HP as part of a plant modification (BDC 92-0220-0). The overload trip condition was caused by concurrently running the new fan motor and an old fan (TEA-FN-1A) motor while both were still powered from the same MCC. Due to E-MC-3A loading limitations, later steps in the modification installation sequence provided for the relocation of the TEA-FN-1A motor power feed to another MCC. The field engineer did not recognize that the modification sequence of running the old and the new fan motors concurrently on the same MCC would cause an overload of the MCC.

Problem Evaluation Report (PER) 294-0324 was initiated following the loss of E-MC-3A and an Incident Review Board (IRB) was convened to investigate the event. As an immediate action, an Engineering "Time Out" was taken on April 27, 1994

to review ongoing design changes with Project Engineers, System Engineers, Design Engineers, and Operations personnel to ensure adequate implementation and test planning. A formal root cause analysis was subsequently performed for the PER and a corrective action plan was developed to preclude the recurrence of a similar event.

3. Following the scram, investigation of the observed reactor power fluctuations showed that they were not the result of reactor core instability or oscillations. The power fluctuations were caused by a degraded position transmitter (RRC-POT-26A) for a Reactor Recirculation (RRC) System flow control valve (FCV) (RRC-FCV-60A). The position transmitter produced slight perturbations in the output signal that caused small step changes in actual FCV position. These valve position changes caused corresponding changes in reactor core flow and power.

When the FCV was in operation at rated power and flow conditions, slight perturbations in the position transmitter output signal do not result in significant changes in reactor core flow because the valve is near full open. However, these same transmitter output perturbations can cause relatively significant changes in reactor core flow when the FCV is near its lower limit such as the 55% flow condition that existed during this event.

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4. Indications of RRC-FCV-60A position fluctuations were previously identified on August 16, 1993. As a result of the investigation, a Work Order was initiated and scheduled to replace RRC-FCV-60A position transmitter RRC-POT-26A. As an interim measure until transmitter replacement, the system engineer provided recommendations in an Interoffice Memorandum (IOM) for operating RRC-FCV-60A in the degraded condition. One of the recommendations was for the FCV hydraulic power unit (HPU) to be shutdown after the valve was placed in the desired position. This action establishes a hydraulic "lock up" of the valve and prevents the valve operator from moving the valve. Operating Procedure PPM 2.2.1, "Reactor Recirculation System," was revised to incorporate the recommendations for interim operation of RRC-FCV-60A; however, the procedure revision did not provide criteria for determining when to "lock up" the FCV, other than to say: "after the FCV is in the desired position... ."

5. RRC-FCV-60A had been maintained in a "lock up" condition most of the time, except during plant maneuvering, since August 1993. However, approximately two weeks before this event, maintenance activities began to replace control rod solenoid scram pilot valves (SSPVs). With the plant power maneuvering requirements necessary to support this maintenance effort, Operations crew management decided not to "lock up" RRC-FCV-60A after each power maneuver. During this two week time period, the FCV was observed to have been operating normally, with no indications of spurious valve position changes.

After reducing reactor power to 50% and core flow to 55% (and having taken the FCV out of "lock up") in response to the E-MC-3A outage and loss of the LP feedwater heaters, Operations crew management decided not to "lock up" RRC-FCV-60A until the plant was returned to the power and flow conditions that existed prior to the event. They expected to promptly recover from the MCC outage, restore the lost heaters, and increase reactor power and flow to the previous values of approximately 70% and 100%, respectively; then they would "lock up" the FCV. The Supply System believes that the decision not to "lock up" RRC-FCV-60A was consistent with procedure provisions. However, this event did reveal that there was insufficient criteria in the governing procedure for determining when to "lock up" the FCV.

The action to manually scram the reactor was consistent with Supply System expectations as conveyed to the plant staff through procedures and training. However, for instances such as in this event, where there are no clear indications that reactor core oscillations are occurring, the Supply System has concluded that refined guidance and training may be prudent to assure additional confirmatory information is taken into consideration prior to scrambling the reactor. Efforts are underway to evaluate and implement, if appropriate, additional guidance and training for unexplained power oscillations.

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Root Cause

The primary root cause for this event (manual scram) was the degraded condition of FCV position transmitter RRC-POT-26A. A secondary root cause was insufficient procedural criteria for determining when to "lock up" the degraded FCV.

Further Corrective Action

1. Replacement of RRC-FCV-60A and 60B position transmitters will be completed prior to plant startup from the Spring 1994 Refueling Outage (R9).
2. Calibration and testing of the replacement FCV position transmitters will be completed prior to plant startup from R9.
3. This Licensee Event Report (LER) will be required to be read by WNP-2 licensed operators prior to plant startup from R9.

Safety Significance

A manual reactor scram is the required immediate action in response to unexplained observed power oscillations. Although an actual RPV low level condition did exist when the water level decreased to -15.4 inches following the reactor scram, the transient was well within the bounds of the WNP-2 safety analysis. This event posed no threat to the health and safety of either the public or plant personnel.

Similar Events

LERs 89-031 and 93-002 reported events where degraded RRC flow control valve system controls contributed to reactor scrams. The degraded system controls in these previous events involved inappropriate setpoints and the negative effects of component interactions following system design changes, modifications, and maintenance. The degraded system controls were not attributed to control circuit or component failures. Thus, these previous event LERs did not include corrective actions that would be expected to prevent the conditions described in this LER.

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EIIS Information

Text Reference EIIS Reference
System Component

Reactor Recirculation (RRC) System AD ---
Flow Control Valve (RRC-FCV-60A) AD FCV
Position Transmitter (RRC-POT-26A) AD ZT
Load Center Breaker (E-CB-31/3) EC BKR (52)
Motor Control Center (E-MC-3A) EC MCC
Reactor Protection System (RPS) JC ---

Average Power Range Monitors (APRMs) JC MON
Reactor Feedwater (RFW) System SJ ---
Low Pressure (LP) Feedwater Heaters SM HX
Bleed Steam (BS) Dump Valve SM FSV
Turbine Non-Return Valve SM FSV2

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM
P.O. Box 98 o George Washington Way o Richland, Washington 99352

May 26, 1994
G02-94-125

Docket No. 50-397

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21
LICENSEE EVENT REPORT NO. 94-008-00

Transmitted herewith is Licensee Event Report No. 94-008-00 for the WNP-2 Plant. This report is submitted in response to the reporting requirements of 10CFR50.73 and discusses the items of reportability, corrective action taken, and action taken to preclude recurrence.

Should you have any questions or desire additional information, please call me or D.A. Swank at (509) 377-4563.

Sincerely

J. V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

JVP/CDM/my
Enclosure

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